UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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December 22, 1980

TO ALL LICENSEES OF OPERATING PLANTS AND
APPLICANTS FOR OPERATING LICENSES AND
HOLDERS OF CONSTRUCTION PERMITS*

Gentlemen:

Subject: Control of Heavy Loads

In January 1978, the NRC published NUREG-0410 entitled, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants - Report to Congress." As part of this program, the Task Action Plan for Unresolved Safety Issue Task No. A-36, "Control of Heavy Loads Near Spent Fuel," was issued.

We have completed our review of load handling operations at nuclear power plants. A report describing the results of this review has been issued as NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants - Resolution of TAP A-36." This report contains several recommendations to be implemented by all licensees and applicants to ensure the safe handling of heavy loads.

The purpose of this letter is to request that you review your controls for the handling of heavy loads to determine the extent to which the guidelines of Enclosure 1 are presently satisfied at your facility, and to identify the changes and modifications that would be required in order to fully satisfy these guidelines.

To expedite your compliance with this request, we have enclosed the following:

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" (Enclosure 1).

Staff Position - Interim Actions for Control of Heavy Loads (Enclosure 2).

Request for Additional Information on Control of Heavy Loads (Enclosure 3).

*With the exception of licensees for Indian Point 2 and 3, Zion 1 and 2 and Three Mile Island 1 (These were previously sent a letter)

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You are requested to implement the interim actions described in Enclosure 2 as soon as possible but no later than 90 days from the date of this letter.

In order to enable the NRC to determine whether operating licenses should be modified (10 CFR 50.54(f)), operating reactor licensees are requested to provide the following:

1. Submit a report documenting the results of your review and the required changes and modifications. This report should include the information identified in Sections 2.1 through 2.4 of Enclosure 3, on how the guidelines of NUREG-0612 will be satisfied. This report should be submitted in two parts according to the following schedule:

   - Submit the Section 2.1 information within six months from the date of this letter.

   - Submit the Sections 2.2, 2.3 and 2.4 information within nine months.

2. Furnish confirmation within six months that implementation of those changes and modifications you find are necessary will commence as soon as possible without waiting staff review, so that all such changes, beyond the above interim actions, will be completed within two years of submittal of Section 2.4 for the above report.

3. Furnish justification within six months for any changes or modifications that would be required to fully satisfy the guidelines of Enclosure 1 which you believe are not necessary.

The criteria in NUREG-0612 are also applicable to applicants for operating licenses. Such applicants are expected to provide the information requested by item 1 above and to meet the same schedule of implementation as indicated in 2 above. Any item for which the implementation date is prior to the expected date of issuance of an operating license will be considered to be a prerequisite to obtaining that license.

For any date that cannot be met, furnish a proposed revised date, justification for the delay, and any planned compensating safety actions during the interim.

This request for information was approved by GAO under a blanket clearance number R0072 which expires November 30, 1983. Comments on burden and duplication may be directed to the U.S. General Accounting Office, Regulatory Reports Review, Room 5106, 441 G Street, N.W., Washington, D.C. 20548.

Sincerely,

Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing

Enclosures:
1. NUREG-0612
2. Staff Position
3. Request for Additional Information

cc: w/o Enclosure (1)
Service List

ENCLOSURE 2

STAFF POSITION - INTERIM ACTIONS FOR CONTROL OF HEAVY LOADS

(1) Safe load paths should be defined per the guidelines of Section 5.1.1(1) (See Enclosure 1);

(2) Procedures should be developed and implemented per the guidelines of Section 5.1.1(2) (See Enclosure 1);

(3) Crane operators should be trained, qualified and conduct themselves per the guidelines of Section 5.1.1(3) (See Enclosure 1);

(4) Cranes should be inspected, tested, and maintained in accordance with the guidelines of Section 5.1.1(6) (See Enclosure 1); and

(5) In addition to the above, special attention should be given to procedures, equipment, and personnel for the handling of heavy loads over the core, such as vessel internals or vessel inspection tools. This special review should include the following for these loads: (1) review of procedures for installation of rigging or lifting devices and movement of the load to assure that sufficient detail is provided and that instructions are clear and concise; (2) visual inspections of load bearing components of cranes, slings, and special lifting devices to identify flaws or deficiencies that could lead to failure of the component; (3) appropriate repair and replacement of defective components; and (4) verify that the crane operators have been properly trained and are familiar with specific procedures used in handling these loads, e.g., hand signals, conduct of operations, and content of procedures.

REQUEST FOR ADDITIONAL INFORMATION ON CONTROL OF HEAVY LOADS

1. INTRODUCTION

Verification by the licensee that the risk associated with load-handling failures at nuclear power plants is extremely low will require a systematic evaluation of all load-handling systems at each site. The following specific information requests have been organized to support such a systematic approach, and provide a basis for the staff's review of the licensee's evaluation. Additionally, they have been organized to address separately the two hazards requiring investigation (i.e., radiological consequences of damage to fuel and unavailability consequences of damage to certain systems). The following general information is provided to assist in this evaluation and reduce the need for clarification as to the intent and expected results of this inquiry.

1. Risk reduction can be demonstrated by either of two approaches:

   a. The likelihood of failure is made extremely low through enhanced handling-system design features (NUREG 0612, Section 5.1.6).

   b. The consequences of a failure can be shown to be acceptable (NUREG 0612, Section 5.1, Criteria I-IV).

   Regardless of the approach selected, the general guidelines of NUREG 0612, Section 5.1.1, should be satisfied to provide maximum practical defense-in-depth.

2. Evaluations concerning radiological consequences or criticality safety, where used, can rely on either the adoption of generic analyses reported in NUREG 0612, requiring only verfication that these generic assumptions are valid for a specific site or employ a site-specific analysis.

3. Systems required for safe shutdown and continued decay heat removal are site-specific and are not, therefore, identified in this request. Individual plants should consider systems and components identified in Regulatory Guide 1.29, Position C.1 (except those systems or portions of systems that are required solely for (a) emergency core cooling, (b) post-accident containment heat removal, or (c) post-accident containment atmosphere cleanup), for evaluation and recognize that the approach taken in this respect is similar to that identified in Regulatory Guide 1.29, Position C.2. The fact that a load-handling system may be prevented from operating during plant conditions requiring the actual or potential use of some of these systems, is rec-

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ognized in this request for information.

4. The scope of this systematic review should include all heavy loads carried in areas where the potential for non-compliance with the acceptance criteria (NUREG 0612, Section 5.1) exists. A summary of typical loads to be considered has been provided in NUREG 0612, Table 3.1-1. It is recognized that some cranes will carry additional miscellaneous loads, some of which are not identifiable in detail in advance. In such cases an evaluation or analysis demonstrating the acceptability of the handling of a range of loads should be provided.

5. At some sites loads which must be evaluated will include licensed shipping casks provided for the transportation of irradiated fuel, solidified radioactive waste, spent resins, or other byproduct material. Licensing under 10CFR71 is not evidence that lifting devices for these shipping casks meet the criteria specified in NUREG 0612, Sections 5.1.1(4), 5.1 1(5), 5.1.6(1), or 5.1.6(3), as appropriate, and thus does not eliminate the need to provide appropriate information concerning these devices. A tabulation (Attachment 5) is provided to indicate multiple-site use of these shipping casks.

The results of the licensee's evaluation, as reported in response to this request, should provide information sufficient for the staff to conduct an independent review to determine that the intent of this effort (i.e., the uniform reduction of the potential hazard from load-handling-system failures) has been satisfied.

2. INFORMATION REQUESTED FROM THE LICENSEE

2.1 GENERAL REQUIREMENTS FOR OVERHEAD HANDLING SYSTEMS

NUREG 0612, Section 5.1.1, identifies several general guidelines related to the design and operation of overhead load-handling systems in the areas where spent fuel is stored, in the vicinity of the reactor core, and in other areas of the plant where a load drop could result in damage to equipment required for safe shutdown or decay heat removal. Information provided in response to this section should identify the extent of potentially hazardous load-handling operations at a site and the extent of conformance to appropriate load-handling guidance.

1. Report the results of your review of plant arrangements to identify all overhead handling systems from which a load drop may result in damage to any system required for plant shutdown or decay heat removal (taking no credit for any

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interlocks, technical specifications, operating procedures, or detailed structural analysis).

2. Justify the exclusion of any overhead handling system from the above category by verifying that there is sufficient physical separation from any load-impact point and any safety-related component to permit a determination by inspection that no heavy load drop can result in damage to any system or component required for plant shutdown or decay heat removal.

3. With respect to the design and operation of heavy-load-handling systems in the reactor building and those load-handling systems identified in 2.1-1, above, provide your evaluation concerning compliance with the guidelines of NUREG 0612, Section 5.1.1. The following specific information should be included in your reply:

   a. Drawings or sketches sufficient to clearly identify the location of safe load paths, spent fuel, and safety-related equipment.

   b. A discussion of measures taken to ensure that load-handling operations remain within safe load paths, including procedures, if any, for deviation from these paths.

   c. A tabulation of heavy loads to be handled by each crane which includes the load identification, load weight, its designated lifting device, and verification that the handling of such load is governed by a written procedure containing, as a minimum, the information identified in NUREG 0612, Section 5.1.1(2).

   d. Verification that lifting devices identified in 2.1.3-c, above, comply with the requirements of ANSI N14.6-1978, or ANSI B30.9-1971 as appropriate. For lifting devices where these standards, as supplemented by NUREG 0612, Section 5.1.1(4) or 5.1.1(5), are not met, describe any proposed alternatives and demonstrate their equivalency in terms of load-handling reliability.

   e. Verification that ANSI B30.2-1976, Chapter 2-2, has been invoked with respect to crane inspection, testing, and maintenance. Where any exception is taken to this standard, sufficient information should be provided to demonstrate the equivalency of proposed alternatives.

   f. Verification that crane design complies with the guidelines of CMAA Specification 70 and Chapter 2-1 of ANSI B30.2-1976, including the demonstration of equivalency of actual design requirements for instances where specific compliance with these standards is not provided.

6. Exceptions, if any, taken to ANSI B30.2-1976 with respect to operator training, qualification, and conduct.

2.2 SPECIFIC REQUIREMENTS FOR OVERHEAD HANDLING SYSTEMS OPERATING IN THE REACTOR BUILDING

NUREG 0612, Section 5.1.4, provides guidelines concerning the design and operation of load-handling systems in the vicinity of spent fuel in the reactor vessel or in storage. Information provided in response to this section should demonstrate that adequate measures have been taken to ensure that, in this area, either the likelihood of a load drop which might damage spent fuel is extremely small, or that the estimated consequences of such a drop will not exceed the limits set by the evaluation criteria of NUREG 0612, Section 5.1, Criteria I through III.

1. Identify by name, type, capacity, and equipment designator, any cranes physically capable (i.e., ignoring interlocks, moveable mechanical stops, or operating procedures) of carrying loads over spent fuel in the storage pool or in the reactor vessel.

2. Justify the exclusion of any cranes in this area from the above category by verifying that they are incapable of carrying heavy loads or are permanently prevented from movement of heavy loads over stored fuel or into any location where, following any failure, such load may drop into the reactor vessel or spent fuel storage pool.

3. Identify any cranes listed in 2.2-1, above, which you have evaluated as having sufficient design features to make the likelihood of a load drop extremely small for all loads to be carried and the basis for this evaluation (i.e., complete compliance with NUREG 0612, Section 5.1.6 or partial compliance supplemented by suitable alternative or additional design features). For each crane so evaluated, provide the load-handling-system (i.e., crane-load-combination) information specified in Attachment 1.

4. For cranes identified in 2.2-1, above, not categorized according to 2.2-3, demonstrate that the criteria of NUREG 0612, Section 5.1, are satisfied. Compliance with Criterion IV will be demonstrated in response to Section 2.4 of this request. With respect to Criteria I through III, provide a discussion of your evaluation of crane operation in the Reactor Building and your determination of compliance. This response should include the following information for each crane:

   a. Where reliance is placed on the installation and use of electrical interlocks or mechanical stops, indicate the circumstances under which these protective devices can be removed or bypassed and the administrative procedures invoked to ensure proper authorization of such action. Discuss any related or proposed technical specifications concerning the bypass of such interlocks.

b. Where reliance is placed on the operation of the Standby Gas Treatment System, discuss present and/or proposed technical specifications and administrative or physical controls provided to ensure that these assumptions remain valid.

c. Where reliance is placed on other site-specific considerations (e.g., refueling sequencing), provide present or proposed technical specifications, and discuss administrative or physical controls provided to ensure the validity of such considerations.

d. Analyses performed to demonstrate compliance with Criteria I through III should conform to the guidelines of NUREG 0612, Appendix A. Justify any exception taken to these guidelines, and provide the specific information requested in Attachment 2, 3, or 4, as appropriate, for each analysis performed.

2.3 SPECIFIC REQUIREMENTS FOR OVERHEAD HANDLING SYSTEMS OPERATING IN PLANT AREAS CONTAINING EQUIPMENT REQUIRED FOR REACTOR SHUTDOWN, DECAY HEAT REMOVAL, OR SPENT FUEL POOL COOLING

NUREG 0612, Section 5.1.5, provides guidelines concerning the design and operation of load-handling systems in the vicinity of equipment or components required for safe reactor shutdown and decay heat removal. Information provided in response to this section should be sufficient to demonstrate that adequate measures have been taken to ensure that in these areas, either the likelihood of a load drop which might prevent safe reactor shutdown or prohibit continued decay heat removal is extremely small, or that damage to such equipment from load drops will be limited in order not to result in the loss of these safety-related functions. Cranes which must be evaluated in this section have been previously identified in your response to 2.1-1, and their loads in your response to 2.1-3-c.

1. Identify any cranes listed in 2.1-1, above, which you have evaluated as having sufficient design features to make the likelihood of a load drop extremely small for all loads to be carried and the basis for this evaluation (i.e., complete compliance with NUREG 0612, Section 5.1.6, or partial compliance supplemented by suitable alternative or additional design features). For each crane so evaluated, provide the load-handling-system (i.e., crane-load-combination) information specified in Attachment 1.

2. For any cranes identified in 2.1-1 not designated as single-failure-proof in 2.3-1, a comprehensive hazard evaluation should be provided which includes the following information:

   a. The presentation in a matrix format of all heavy loads and potential impact areas where damage might occur to safety-related equipment. Heavy loads identification should include designation and weight or cross-reference to information provided in 2.1-3-c. Impact areas should be identified by construction zones and elevations or by some other method such that the impact area can be located on the plant general arrangement drawings. Figure 1 provides a typical matrix.

   b. For each interaction identified, indicate which of the load and impact area combinations can be eliminated because of separation and redundancy of safety-related equipment, mechanical stops and/or electrical interlocks, or other site-specific considerations. Elimination on the basis of the aforementioned consideration should be supplemented by the following specific information:

      (1) For load/target combinations eliminated because of separation and redundancy of safety-related equipment, discuss the basis for determining that load drops will not affect continued system operation (i.e., the ability of the system to perform its safety-related function).

      (2) Where mechanical stops or electrical interlocks are to be provided, present details showing the areas where crane travel will be prohibited. Additionally, provide a discussion concerning the procedures that are to be used for authorizing the bypassing of interlocks or removable stops, for verifying that interlocks are functional prior to crane use, and for verifying that interlocks are restored to operability after operations which require bypassing have been completed.

      (3) Where load/target combinations are eliminated on the basis of other, site-specific considerations (e.g., maintenance sequencing), provide present and/or proposed technical specifications and discuss administrative procedures or physical constraints invoked to ensure the validity of such considerations.

c. For interactions not eliminated by the analysis of 2.3-2-b, above, identify any handling systems for specific loads which you have evaluated as having sufficient design features to make the likelihood of a load drop extremely small and the basis for this evaluation (i.e., complete compliance with NUREG 0612, Section 5.1.6, or partial compliance supplemented by suitable alternative or additional design features). For each so evaluated, provide the load-handling-system (i.e., crane-load-combination) information specified in Attachment 1.

d. For interactions not eliminated in 2.3-2-b or 2.3-2-c, above, demonstrate using appropriate analysis that damage would not preclude operation of sufficient equipment to allow the system to perform its safety function following a load drop (NUREG 0612, Section 5.1, Criterion IV). For each analysis so conducted, the following information should be provided:

   (1) An indication of whether or not, for the specific load being investigated, the overhead crane-handling system is designed and constructed such that the hoisting system will retain its load in the event of seismic accelerations equivalent to those of a safe shutdown earthquake (SSE).

   (2) The basis for any exceptions taken to the analytical guidelines of NUREG 0612, Appendix A.

   (3) The information requested in Attachment 4.

NOTES TO FIGURE 1

Note 1: Indicate by symbols the safety-related equipment. The licensee should provide a list consistent with the clarification provided in 1.2-3.

Note 2: Hazard Elimination Categories

a. Crane travel for this area/load combination prohibited by electrical interlocks or mechanical stops.

b. System redundancy and separation precludes loss of capability of system to perform its safety-related function following this load drop in this area.

c. Site-specific considerations eliminate the need to consider load/equipment combination.

d. Likelihood of handling system failure for this load is extremely small (i.e. section 5.1.6 NUREG 0612 satisfied).

e. Analysis demonstrates that crane failure and load drop will not damage safety-related equipment.

FIGURE 1

Typical Load/Impact Area Matrix

CRANE: (IDENTIFY THE CRANE BY NAME AND EQUIPMENT NUMBER)

LOCATION	INDICATE THE BUILDING(S) CORRESPONDING TO THE IMPACT AREA(S) EXAMPLE: REACTOR BUILDING, AUXILIARY BUILDING					
IMPACT AREA / LOADS	(IDENTIFY AREA BY CONSTRUCTION ZONES) Example: Column Line F-S, Column Line B9-B12					
	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
(Heavy Load identification should include designation and weight) Example Spent Fuel Cask HLI 10/24 (100 tons)	(Indicate the various elevations) Example: Elev. 435'	Note 1	Note 2			

SINGLE-FAILURE-PROOF HANDLING SYSTEMS

1. Provide the name of the manufacturer and the design-rated load (DRL). If the maximum critical load (MCL), as defined in NUREG 0554, is not the same as the DRL, provide this capacity.

2. Provide a detailed evaluation of the overhead handling system with respect to the features of design, fabrication, inspection, testing, and operation as delineated in NUREG 0554 and supplemented by the identified alternatives specified in NUREG 0612, Appendix C. This evaluation must include a point-by-point comparison for each section of NUREG 0554. If the alternatives of NUREG 0612, Appendix C, are used for certain applications in lieu of complying with the recommendation of NUREG 0554, this should be explicitly stated. If an alternative to any of those contained in NUREG 0554 or NUREG 0612, Appendix C, is proposed, details must be provided on the proposed alternative to demonstrate its equivalency.[1/]

3. With respect to the seismic analysis employed to demonstrate that the overhead handling system can retain the load during a seismic event equal to a safe shutdown earthquake, provide a description of the method of analysis, the assumptions used, and the mathematical model evaluated in the analysis. The description of assumptions should include the basis for selection of trolley and load position.

4. Provide an evaluation of the lifting devices for each single-failure-proof handling system with respect to the guidelines of NUREG 0612, Section 5.1.6.

5. Provide an evaluation of the interfacing lift points with respect to the guidelines of NUREG 0612, Section 5.1.6.

1/ If the crane in question has previously been approved by the staff as satisfying NUREG 0554, Reg. Guide 1.104, or Part B to BTP-ASB9-1, please reference the date of the staff's safety evaluation report or approval letter in lieu of providing the information requested by item 2.

ANALYSIS OF RADIOLOGICAL RELEASES

The following information should be provided for an analysis conducted to demonstrate compliance with Criterion I of NUREG 0612, Section 5.1.

1. INITIAL CONDITIONS/ASSUMPTIONS

   a. Identify the time after shutdown, the number of fuel assemblies damaged, and the assumed duration of radiological release associated with each accident analyzed.

   b. NUREG 0612, Table 2.1-2, provides the assumptions used to arrive at generic conclusions concerning radiological dose consequences. To rely on the radiological dose analysis of NUREG 0612, the licensee should verify that these assumptions are conservative with regard to the plant/site evaluated. If the assumptions are not conservative for the specific plant, or if a more site-specific analysis is required, the licensee should identify plant-specific assumptions used in place of those tabulated.

   c. Identify and provide the basis (e.g., USNRC Regulatory Guide 1.25) for any assumptions employed in site-specific analyses not identified in NUREG 0612, Table 2.1-2.

   d. Dose calculations based on the termination or mitigation of radiological releases should be supported by information sufficient to demonstrate both that the time delay assumed is conservative and that the system provided to accomplish such termination or mitigation will perform its safety function upon demand (i.e., the system meets the criteria for an Engineered Safety Feature). Specific information so provided should include the following:

      (1) Details concerning the location of accident sensors, parameters monitored and the values of these parameters at which a safety signal will be initiated, system response time (including valve-operation time), and the total time required to automatically shift from normal operation to isolation or filtration following an accident.

      (2) A description of the instrumentation and controls associated with the Engineered Safety Feature which includes information sufficient to demonstrate that the requirements (Section 4) of IEEE 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," are satisfied.

(3) A description of any Engineered Safety Feature filter system which includes information sufficient to demonstrate compliance with the guidelines of USNRC Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants."

(4) A discussion of any initial conditions (e.g., manual valves locked shut, containment airlocks or equipment hatches shut) necessary to ensure that releases will be terminated or mitigated upon Engineered Safety Feature actuation and the measures employed (i.e., Technical Specification and administrative controls) to ensure that these initial conditions are satisfied and that Engineered Safety Feature systems are operable prior to the load lift.

2. METHOD OF ANALYSIS

Discuss the method of analysis used to demonstrate that post-accident dose will be well within 10CFR100 limits. In presenting methodology used in determining the radiological consequences, the following information should be provided.

a. A description of the mathematical or physical model employed.

b. An identification and summary of any computer program used in this analysis.

c. The consideration of uncertainties in calculational methods, equipment performance, instrumentation response characteristics, or other indeterminate effects taken into account in the evaluation of the results.

3. CONCLUSION

Provide an evaluation comparing the results of the analysis to Criterion I of NUREG 0612, Section 5.1. If the postulated heavy-load-drop accident analyzed bounds other postulated heavy-load drops, a list of these bounded heavy loads should be provided.

CRITICALITY ANALYSIS

The following information should be provided for analysis conducted to demonstrate compliance with Criterion II of NUREG 0612, Section 5.1

1. INITIAL CONDITIONS/ASSUMPTIONS

The conclusions of NUREG 0612, Section 2.2, are based on a particular model fuel assembly. If a licensee uses the results of Section 2.2 rather than performing an independent neutronics analysis, the assumptions should be verified to be compatible with plant-specific design. For any analysis conducted, the following assumptions should be provided as a minimum:

a. Water/UO_2 volume ratio
b. The boron concentration for the refueling water and spent-fuel pool
c. The amount of neutron poison in the fuel
d. Fuel enrichment
e. The reactivity insertion value due to crushing of the core
f. The k_{eff} value allowed by technical specifications for the core during refueling

2. METHOD OF ANALYSIS

Provide the method of analysis used to demonstrate that accidental dropping of a heavy load does not result in a configuration of the fuel such that k_{eff} is larger than 0.95. The discussion of the method of analysis should include the following information:

a. Identification of the computer codes employed
b. A discussion of allowances or compensation for calculation and physical uncertainties

3. CONCLUSION

Provide an evaluation comparing the results of the analysis to Criterion II of NUREG 0612, Section 5.1. If the postulated heavy-load-drop accident bounds other postulated heavy-load drops, a list of these bounded heavy loads should be provided.

Attachment (4)

ANALYSIS OF PLANT STRUCTURES

The following information should be provided for analyses conducted to demonstrate compliance with Criteria III and IV of NUREG 0612, Section 5.1.

1. INITIAL CONDITIONS/ASSUMPTIONS

   Discuss the assumptions used in the analysis, including:

   a. Weight of heavy load
   b. Impact area of load
   c. Drop height
   d. Drop location
   e. Assumptions regarding credit taken in the analysis for the action of impact limiters
   f. Thickness of walls or floor slabs impacted
   g. Assumptions regarding drag forces caused by the environment
   h. Load combinations considered
   i. Material properties of steel and concrete

2. METHOD OF ANALYSIS

   Provide the method of analysis used to demonstrate that sufficient load-carrying capability exists within the wall(s) or floor slab(s). Identify any computer codes employed, and provide a description of their capabilities. If test data was employed, provide it and describe its applicability.

3. CONCLUSION

   Provide an evaluation comparing the results of this analysis with Criteria III and IV of NUREG 0612, Section 5.1. Where safe-shutdown equipment has a ceiling or wall separating it from an overhead handling system, provide an evaluation to demonstrate that postulated load drops do not penetrate the ceiling or cause secondary missiles that could prevent a safe-shutdown system from performing its safety function.

SHIELDED SHIPPING CASKS CERTIFICATED FOR NUCLEAR POWER PLANTS

I - Fuel (New and Spent)

CERT.	MODEL	PRIMARY LICENSEE	GROSS LOT IN LBS. (APPROX.)	SECONDARY LICENSEE*
4986	RA-1, 2, 3, J	General Electric Co.		TVA
5450	RCC, 1, 2, 3	Westinghouse Electric		VEP, DLC
5805	Vandenburgh	Chem-Nuclear Systems, Inc.	70,000	APC, CPL, DLP, DPC, FPL, FPC, JCP, NPP, VEP
5901	NFS Model 100	Nuclear Fuel Services	126,200	CPC, PGE
5938	HN[illegible]F		48,000	PEC
6078	927A1 927C1	Combustion Engineering, Co.	6200 7000	APL
6206	B	Babcock & Wilcox Co.	6940	DPC, FPC
6273	48 (Series)		4500	VEP
6375	PB-1	Chem-Nuclear Systems, Inc.	67,050	APC, BEC, CPL, DPC FPL, FPC, GPC, JCP, MYA, MEC, NNE, NSP, PNY, TVA, VEP
6400	Super Tiger	Westinghouse Electric Co.	45,000	APL, CPC, DLP, DLC, MEC, NPP, SMU, VEP
6698	NFS-4	Nuclear Fuel Services, Inc.	50,000	BGE, BEC, CWE, DLP, DPC, FPL, FPC, JCP, MYA, RGE, SCE, WMP,
9001	IF 300	General Electic Co.	140,000	CPL, CWE
9010	NLI-1/2	NL Industries, Inc.	47,500	BEC, FPL, VYC
9044	GE-1600	General Electric Co.	23,000	APC, BGE, BEC, CPL, CPC, DPC, FPL, FPC, GPC, IEL, JCP, MEC, NNE, NSP, VEP, VYC

*See attached list of abbreviations.

SHIELDED SHIPPING CASKS CERTIFICATED FOR NUCLEAR POWER PLANTS

II - Waste

CERT.	MODEL	PRIMARY LICENSEE	GROSS LOT IN LBS. (APPROX.)	SECONDARY LICENSEE*
5026	BC-48-220	Chem-Nuclear Systems, Inc.	71,000	APC, BEC, CPL, CWE, CYA, DPC, DLC, FPL, FPC, JCP, NPP, VEP, WPS
6058	B3-1	Nuclear Engineering Co.	30,000	APL, CPC, DLP, IEL, MEC, NPP, NSP, PGE, SMU, TEC, VEP
6144	6144	Nuclear Engineering Co.	42,000	APC, APL, CPL, CEC, CPC, DLP, DPC, FPL, FPC, GPC, IEL, JCP, MEC, NPP, NSP, PGE, PNY, RGE, SMU, VEP
6244	6244	Chem-Nuclear Systems, Inc.	46,000	APC, CPL, CWE, DPC, FPL, FPC, GPC, JCP, MEC, NMP, NSP, PEC, VEP, WMP
6272	Poly Panther	Nuclear Engineering Co.	6100	APL, CPC, DLP, MEC NPP, SMU, VEP
6568	LL-60-150	Tennessee Valley Auth.	73,000	
6574	HN 200	Hittman Nuclear and Development Corp.	47,000	APL, BGE, CWE, CEC, DLP, DLC, IME, JCP, MYA, MEC, NPP, PEC, PNY, VYC, YAC
6601	LL-50-100	Chem-Nuclear Systems, Inc.	70,000	APC, BEC, CPL, CYA, CEC, CPC, DLP, DPC, FPL, FPC, JCP, NPP, NNE, PEG, PGE, TVA, VEP
6679	1/2 Super Tiger	Nuclear Engineering Co.	45,000	APL, CPC, DLP, MEC, NPP, SMU, VEP
6722	BS-33-150	Tennessee Valley Auth.	51,000	

*See attached list of abbreviations.

SHIELDED SHIPPING CASKS CERTIFICATED FOR NUCLEAR POWER PLANTS

II - Waste

CERT.	MODEL	PRIMARY LICENSEE	GROSS LOT IN LBS. (APPROX.)	SECONDARY LICENSEE*
6744	Poly Tiger	Nuclear Engineering Co.	35,000	APL, BEC, CPC, DLP, MEC, NPP, SMU, VEP
6771	SN-1	Nuclear Engineering Co.	60,000	APL, CPC, DLP, NPP, SMU, VEP
9074	AP-100		28,000	DLC
9079	HN-100 Ser. 2	Hittman Nuclear and Development Corp.	98,000	APL, BGE, CEC, CWE, DLP, IME, JCP, MYA, MEC, NPP, PEC
9080	HN-600	Hittman Nuclear and Development Corp.	42,000	BGE, CWE, CEC, DLP, IME, IEL, JCP, MYA, MEC, NPP, PEC, YAC
9086	HN-100 Ser. 1	Hittman Nuclear and Development Corp.	46,000	APL, BGE, CWE, DLP, IME, JCP, MYA, MEC, NPP, NNE, PEC, RGE, VYC
9089	HN-100S	Hittman Nuclear and Development Corp.	36,500	BGE, CWE, CEC, IME, JCP, MYA, NPP, PEC
9092	HN-300	Hittman Nuclear and Development Corp.	43,000	MYA
9093	HN-400	Hittman Nuclear and Development Corp.	43,000	MYA
9094	CNSI-14-195-H	Chem-Nuclear Systems, Inc.	56,500	APC, APL, BEC, CPL, CWE, CYA, CEC, CPC, DPC, FPL, FPC, GPC, JCP, MEC, NMP, NNE, NSP, OPP, PGE, PEC, PGC, PNY, PEG, TVA, VEP
9096	CNSI-21-300	Chem-Nuclear Systems, Inc.	57,450	APC, APL, CPL, CEC, DPC, FPL, FPC, GPC, JCP, MEC, NMP, NNE, PNY, PEG, VEP

*See attached list of abbreviations.

SHIELDED SHIPPING CASKS CERTIFICATED FOR NUCLEAR POWER PLANTS

II - Waste

CERT.	MODEL	PRIMARY LICENSEE	GROSS LOT IN LBS. (APPROX.)	SECONDARY LICENSEE*
9105	RAD-Waste CR.I	Chem-Nuclear Systems, Inc.	58,400	APC, CPL, DPC, FPL, FPC, GPC, JCP, MEC, NMP, VEP
9108	AL-33-90	Chem-Nuclear Systems, Inc.	41,300	APC, CPL, CWE, CEC, DPC, FPL, FPC, JCP, NPP, NMP, NNE, PGC, VEP, WEP
9111	CN6-80A	Chem-Nuclear Systems, Inc.	51,500	APC, CPL, CWE, CEC, DPC, FPL, FPC, GPC, MEC, NNE, PGC, SMU, VEP
9113	7-100	Chem-Nuclear Systems, Inc.	7000	APC, BEC, CPL, CWE, CYA, DPC, FPL, FPC, GPC, JCP, MEC, NMP, NNE, NSP, VEP
9122	18-450	Chem-Nuclear Systems, Inc.	61,000	BEC

*See attached list of abbreviations.

SHIELDED SHIPPING CASKS CERTIFICATED FOR NUCLEAR POWER PLANTS

III - Byproducts

CERT.	MODEL	PRIMARY LICENSEE	GROSS LOT IN LBS. (APPROX.)	SECONDARY LICENSEE*
5971	GE-200		10,000	PEC
5980	GE-600		18,500	NNE, NSP
6275	LL-28-4	Chem-Nuclear Systems, Inc.	30,000	APC, CPL, DPC, FPL, FPC, NPP, VEP
9081	CNS-1600	Chem-Nuclear Systems, Inc.	26,000	APC, BGE, CPL, DPC, FPL, FPC, GPC, NSP, TVA, VEP

*See attached list of abbreviations.

LICENSEE ABBREVIATIONS

APC	Alabama Power Company
APL	Arkansas Power and Light Company
BEC	Boston Edison Company
BGE	Baltimore Gas and Electric Company
CEC	Consolidated Edison Company
CPC	Consumers Power Company
CPL	Carolina Power and Light Company
CWE	Commonwealth Edison Company
CYA	Connecticut Yankee Atomic Power Company
DLC	Duquesne Light Company
DLP	Dairyland Power Cooperative
DPC	Duke Power Company
FPC	Florida Power Corporation
FPL	Florida Power and Light Company
GPC	Georgia Power Company
IEL	Iowa Electric Light and Power Company
IME	Indiana and Michigan Electric Company
JCP	Jersey Central Power and Light Company
MEC	Metropolitan Edison Company
MYA	Maine Yankee Atomic Power Company
NMP	Niagara Mohawk Power Corporation
NNE	Northeast Nuclear Energy Company
NPP	Nebraska Public Power Corporation
NSP	Northern States Power Company
OPP	Omaha Public Power District
PEC	Philadelphia Electric Company
PEG	Public Service Electric and Gas Company
PGC	Portland General Electric Company
PNY	Power Authority of the State of New York
RGE	Rochester Gas and Electric Corporation
SMU	Sacramento Municipal Utilities Corporation
TEC	Toledo Edison Company
TVA	Tennessee Valley Authority
VEP	Virginia Electric and Power Company
VYC	Vermont Yankee Nuclear Power Corporation
YAC	Yankee Atomic Electric Company
WMP	Wisconsin-Michigan Power Company
WPS	Wisconsin Public Service Corporation